VERMONT YANKEE
NUCLEAR POWER CORPORATION
185 OLD FERRY ROAD, PO BOX 7002, BRATTLEBORO, VT 05302-7002
(802) 257-5271

August 20, 2001
BVY 01-65

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Subject: **Vermont Yankee Nuclear Power Station**
License No. DPR-28 (Docket No. 50-271)
Technical Specification Proposed Change No. 248
Alternate Train Testing

Pursuant to 10CFR50.90, Vermont Yankee (VY) hereby proposes to amend its Facility Operating License, DPR-28, by incorporating the attached proposed change into the VY Technical Specifications (TS). This proposed change revises certain TS requirements associated with demonstrating the operability of alternate trains when redundant equipment is made or found to be inoperable. These excessive testing requirements are unnecessary to maintain adequate assurance of system operability, and may result in unintended consequences, such as premature wear of components. Other surveillance requirements provide adequate assurance that redundant systems and components are operable and capable of performing their intended safety function.

Attachment 1 to this letter contains supporting information and the safety assessment of the proposed change. Attachment 2 contains the determination of no significant hazards consideration. Attachment 3 provides the marked-up version of the current Technical Specification pages. Attachment 4 is the retyped Technical Specification pages.

VY has reviewed the proposed Technical Specification change in accordance with 10CFR50.92 and concludes that the proposed change does not involve a significant hazards consideration.

VY has also determined that the proposed change satisfies the criteria for a categorical exclusion in accordance with 10CFR51.22(c)(9) and does not require an environmental review. Therefore, pursuant to 10CFR51.22(b), no environmental impact statement or environmental assessment needs to be prepared for this change.

Upon acceptance of this proposed change by the NRC, VY requests that a license amendment be issued no later than six months from the date of this letter for implementation within 60 days of its effective date.

ADD1

If you have any questions on this transmittal, please contact Mr. Gautam Sen at (802) 258-4111.

Sincerely,

VERMONT YANKEE NUCLEAR POWER CORPORATION

Michael A. Balduzzi
Senior Vice President and Chief Nuclear Officer

STATE OF VERMONT)
)ss
WINDHAM COUNTY)

Then personally appeared before me, Michael A. Balduzzi, who, being duly sworn, did state that he is Senior Vice President and Chief Nuclear Officer of Vermont Yankee Nuclear Power Corporation, that he is duly authorized to execute and file the foregoing document in the name and on the behalf of Vermont Yankee Nuclear Power Corporation, and that the statements therein are true to the best of his knowledge and belief.

Sally A. Sandstrum, Notary Public
My Commission Expires February 10, 2003

Attachments

cc: USNRC Region 1 Administrator
USNRC Resident Inspector - VYNPS
USNRC Project Manager - VYNPS
Vermont Department of Public Service

Attachment 1

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 248

Alternate Train Testing

Supporting Information and Safety Assessment of Proposed Change

INTRODUCTION

Purpose

This proposed change revises certain, unnecessary Technical Specification (TS) surveillance requirements (SRs) associated with ensuring the operability of alternate trains when redundant equipment is made or found to be inoperable. Although alternate testing is being eliminated, the basis for assuring operability remains the adequacy of the regular, periodic testing and post-maintenance testing.

Current Technical Specifications

Current VY TS require alternate testing of systems, subsystems, and components for the purpose of demonstrating operability of systems and components as outlined in the following Table 1 when other systems, subsystems, or components are out of service. These tests are currently required to be performed within 24 hours of when a system, subsystem, or component is declared inoperable.

Table 1

System/Component	Current TS
Standby Liquid Control System	4.4.B
Core Spray System	4.5.A.2
Low Pressure Coolant Injection Pumps	4.5.A.3
Low Pressure Coolant Injection System	4.5.A.4
Containment Cooling System	4.5.B.2
Residual Heat Removal Service Water Pumps	4.5.C.2
Residual Heat Removal Service Water System	4.5.C.3
Station Service Water System	4.5.D.2
Alternate Cooling Tower System	4.5.D.3
High Pressure Coolant Injection System	4.5.E.2
Automatic Pressure Relief System	4.5.F.2
Emergency Diesel Generators	4.5.H.1
Standby Gas Treatment System	4.7.B.3.c
Emergency Diesel Generators	4.10.B.1 4.10.B.3.b.2

Description of Proposed Change

The proposed change will eliminate or alter the testing requirements identified in Table 1. Associated changes to the TS Bases are also being made to conform to the changed TS. In addition, two administrative changes are proposed to correct format or typographical errors contained in current TS.

Table 2 provides a detailed description of each proposed change, along with the basis for change and a safety assessment. With these changes, the TS SRs are sufficient to maintain adequate assurance of system operability as needed to mitigate design basis events without reliance on alternate testing.

Need for Change

The proposed change eliminates unnecessary testing of redundant equipment and provides for the increased availability of safety systems, recognizing that equipment trains may be incapable of performing their safety function when removed from service to perform surveillance testing. The changes will reduce excessive testing that could result in undue and premature component wear. Alternate testing can increase the probability of equipment failure, as well as increasing the potential for human error in (1) lining up the system for testing; (2) operating the system to demonstrate operability; and (3) returning the system or subsystem to service in the proper configuration. Testing during Limiting Conditions for Operation may result in operation with a loss of safety function and can also unnecessarily burden operations due to the urgency and importance of meeting surveillance requirements. Therefore, because these requirements result in unnecessary hardships or loss of safety function without a compensating increase in the level of quality or safety, such testing should be discontinued.

SYSTEM DESCRIPTIONS

The following system descriptions provide information on design features that may be unique to the design of VY or helpful in understanding inter-system relationships.

Station Service Water System

The VY Station Service Water (SSW) system is designed to provide water for turbine and reactor auxiliary equipment cooling during normal operation and to provide cooling water in conjunction with the Residual Heat Removal Service Water (RHRSW) pumps for reactor shutdown cooling. Also, the SSW system provides cooling water to systems and equipment required to operate under accident conditions.

The SSW system consists of four pumps, associated valves and piping, one non-essential equipment cooling loop, two essential equipment cooling loops, and additional essential and non-essential equipment cooling loads. Two essential equipment cooling loops provide redundant capability for analyzed accidents and transients. The SSW system is arranged in a dual header system with two SSW pumps on each header. Each header supplies cooling water to a reactor building closed cooling water system heat exchanger, emergency core cooling system room ventilation coolers, a diesel generator cooler, and a set of RHRSW pumps, which supply water to the Residual Heat Removal (RHR) heat exchangers. Two operable SSW pumps with one or both essential equipment cooling loops in operation, provide adequate cooling for analyzed accidents and transients.

RHRSW System

The RHRSW system is designed to provide a source of cooling water for the RHR system during normal shutdown conditions and for the RHR system during a loss of off-site power. The RHRSW pumps are supplied from the SSW system, and the cooling water is then pumped through the RHR heat exchangers and is returned to the SSW system.

Alternate Cooling System

The VY Alternate Cooling System (ACS) (also known as the Alternate Cooling Tower System) is designed to provide the necessary heat sink for normal post-shutdown conditions in the event that the SSW system is unavailable due to a loss of the Vernon Dam with a subsequent loss of the Vernon Pond, flooding of the SSW intake structure, or a fire in the SSW intake structure, which disables all four SSW pumps. This system is not classified as an engineered safety system. The primary source of ACS water consists of a deep basin, located below the west cooling tower. Water from the deep basin is supplied to the suction of the RHRSW pumps and then pumped to the emergency diesel generators, RHR heat exchangers, and small cooling loads, such as the RHRSW pump motor coolers and room coolers. Water is then returned to the cooling tower and latent heat is transferred to the atmosphere. The RHR system will provide cooling as if the plant were being supplied with river water and will function to safely remove the sensible and decay heat from the reactor and other heat loads. Additional details are provided in UFSAR section 10.8.

High Pressure Core Cooling Systems

The High Pressure Coolant Injection (HPCI) system ensures that the reactor core is adequately cooled in the event of a postulated small break loss-of-coolant accident (LOCA). The Reactor Core Isolation Cooling (RCIC) system provides makeup water to the reactor vessel following a reactor vessel isolation when normal sources of feedwater are not available. While not designed or credited as an emergency core cooling system (ECCS), RCIC functions similarly to the HPCI system. Both the HPCI and RCIC systems use a steam-driven turbine-pump to provide water to the reactor vessel. The steam supply for these pumps is the reactor.

The Automatic Depressurization System (ADS) serves as a backup to the HPCI system in the event of a small break LOCA. ADS can reduce the reactor system pressure to allow the low pressure cooling systems to provide water to cool the core. Since the HPCI, RCIC, and ADS systems are high pressure systems, certain tests on these systems require high pressure steam.

Emergency Diesel Generators

VY has two emergency diesel generators (EDGs), either of which is capable of supplying 100% of the minimum emergency safeguards equipment loads under postulated design basis accidents.

SAFETY ASSESSMENT

The alternate testing requirements for demonstrating operability of systems and components were originally included in TS because there was a lack of plant operating history and equipment failure data. However, plant operating history now shows that testing of alternate trains when one train is inoperable is not necessary to provide adequate assurance of system operability. In fact, taking the other train out of service for testing creates the risk of the second system failing. In some instances, the failures may be related to the test itself and not an indication that the train would have failed should it have been needed to actually mitigate an accident. Operability of these systems or components can be assured by performing regular, periodic surveillances and preventive maintenance.

Furthermore, equipment may be incapable of performing its intended safety function during the performance of surveillance testing. This could result in unintended operation with a loss of safety function when the opposite train is already inoperable.

The alternate train testing requirements were included in early TS to provide a positive demonstration that a loss of safety function had not occurred due to a common cause. However, the added assurance of system or component operability that is demonstrated by testing is not sufficient to justify the unintended consequences and potential loss of safety function due to the additional testing. The current boiling water reactor (BWR) Standard Technical Specifications and the TS approved for more recently licensed BWRs accept the logic that testing equipment to demonstrate operability is not required when another subsystem is inoperable (except for diesel generator testing when common cause failure cannot be ruled out). Instead, operability is based on satisfactory performance of regularly-scheduled surveillance, post-maintenance or other specified performance tests. The early NRC staff position for alternate testing has been revised and not included in STS or newly issued TS.

Based on the risk of the redundant subsystem failing, past operational experience, and the similarity to other BWRs, it is acceptable to eliminate requirements for testing alternate systems or components when redundant equipment is made or found to be inoperable.

As stated in NRC Generic Letter 87-09[1], "It is overly conservative to assume that systems or components are inoperable when a surveillance requirement has not been performed. The opposite is in fact the case; the vast majority of surveillance demonstrate the systems or components in fact are operable." Therefore, reliance on the specified surveillance intervals does not result in a reduced level of confidence concerning the equipment availability.

The benefits and drawbacks of alternate train testing have been extensively studied by the nuclear power industry. The benefits of alternate testing primarily relate to the decreased potential for an undetected failure. The drawbacks include:

- Increased system unavailability during testing
- Increased system unavailability due to repair of demand-related and test-related failures
- Reduced reliability due to degradation from testing
- Increased potential of plant transients initiated from surveillance testing
- Increased potential for plant shutdown due to transients resulting from surveillance testing
- Diversion of operations and maintenance personnel for testing
- Potential increase in occupational radiation exposure from surveillance testing

STS and current BWR operating practice accept the philosophy of system operability based on satisfactory performance of periodic, post-maintenance, or other specified performance tests without requiring additional testing when another system or subsystem is made or found to be inoperable. This is also consistent with the guidance provided in GL 93-05[2].

The following Table 2 provides a detailed description of each change, including the basis for the change and a safety assessment. The Change Numbers in the left-hand column correspond to the boxed (!) annotation numbers in Attachment 3, "Marked-Up Version of the Current Technical Specifications."

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[1] NRC Generic Letter 87-09, "Sections 3.0 and 4.0 of the Standard Technical Specifications (STS) on the Applicability of Limiting Conditions for Operation and Surveillance Requirements," June 4, 1987

[2] NRC Generic Letter 93-05, "Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing during Power Operation," September 27, 1993

Table 2

<table>
<tr><th>Change #</th><th>Current Technical Specification</th><th>Proposed Change</th></tr>
<tr><td rowspan="2">1</td><td>CTS 4.4.B requires that when a component in the SLC system becomes inoperable, its redundant component shall be or shall have been demonstrated to be operable within 24 hours.</td><td>Delete CTS 4.4.B and add the word, "Deleted" in its place.</td></tr>
<tr><td colspan="2">Basis / Safety Assessment:

CTS 4.4.B requires that the remaining SLC subsystem be demonstrated to be operable within 24 hours of determining inoperability of the other SLC subsystem. This change precludes performing unnecessary SLC testing and will allow credit to be taken for normal periodic surveillance as a verification of operability and availability of the remaining SLC subsystem. Other SRs (e.g., periodic and post-maintenance testing in accordance with the VY IST program, as required by SR 4.6.E.2) are adequate to ensure system operability.

This change is acceptable because existing SR 4.4.B only provides marginal assurance of system operability, and may in fact have an overall cumulative negative effect on plant safety.</td></tr>
<tr><td rowspan="2">2</td><td>Two typographical errors exist in CTS SR 4.5:

(1) On page 99 of CTS, under the heading, "Applicability," the current first word is misspelled as "Applied."
(2) On page 108 of CTS, SR 4.5.G.1.c is misdesignated as paragraph "c."</td><td>Change the two typographical errors as follows:

(1) On page 99 of CTS, under the heading, "Applicability," change the first word from "Applied" to "Applies."
(2) On page 108 of CTS, SR 4.5.G.1.c is re-designated as SR 4.5.G.1.d.</td></tr>
<tr><td colspan="2">Basis / Safety Assessment:

(1) The typographical error on page 99 dates back to the 1970s. Although it has not been determined when or how this error was introduced (original license was "Applies"), it is clear that this word should be "Applies" to be consistent with or Specifications within the TS, such as TS 3.5.

(2) The typographical error on page 108 of CTS was inadvertently introduced as part of the license amendment request for Amendment No. 195. A review of the re-typed TS pages provided by VY letter BVY 00-83, dated September 19, 2000, indicates this error. Prior to Amendment No. 195, this SR was correctly designated as 4.5.G.1.d. (Note, there was no change to this SR as part of Amendment No. 195.)

These two corrections involve administrative errors that have no bearing on the obvious meaning or intent of the TS. Making these corrections does not change any technical requirements and are acceptable since they merely correct administrative errors introduced through the license amendment process.</td></tr>
</table>

Table 2
(continued)

<table>
<tr><th>Change #</th><th>Current Technical Specification</th><th>Proposed Change</th></tr>
<tr><td rowspan="2">3</td><td>CTS 4.5.A.2 requires that when one core spray subsystem is made or found to be inoperable, the active components of the redundant core spray subsystem shall have been or shall be demonstrated to be operable within 24 hours.</td><td>Delete CTS 4.5.A.2 and add the word, “Deleted” in its place.</td></tr>
<tr><td colspan="2">Basis / Safety Assessment:

The core spray (CS) system is required in conjunction with the LPCI subsystems to provide adequate core cooling to the reactor core in the event of a loss of coolant accident (LOCA). The LPCI subsystems are completely independent of the CS systems; however, LPCI does function with CS to prevent excessive fuel clad temperature in accordance with the requirements of 10CFR50, Appendix K.

Should one CS subsystem become inoperable, the remaining CS subsystem, LPCI subsystems, and associated emergency diesel generators (EDGs) are required to be operable by Limiting Condition for Operation 3.5.A.2. These systems provide extensive margin to that needed to ensure adequate core cooling. CTS 3.5.A.2 only permits continued reactor operation for up to 7 days with one CS subsystem inoperable, provided that the conditions of CTS 3.5.A.2 are met.

Current SR 4.5.A.1, which includes CS testing in accordance with Section XI of the ASME code, provides adequate surveillance to ensure operability of the redundant subsystem. Alternate train testing, which currently must occur within 24 hours of a subsystem being declared inoperable, does not contribute any significant assurance of system operability, and may in fact lead to unintended or premature system failure.

This change is therefore acceptable, since adequate assurance of CS system operational readiness is provided through other SRs.</td></tr>
</table>

Table 2
(continued)

<table>
<tr><th>Change #</th><th>Current Technical Specification</th><th>Proposed Change</th></tr>
<tr><td rowspan="2">4</td><td>CTS 4.5.A.3 requires that when one of the low pressure coolant injection (LPCI) pumps is made or found to be inoperable, the remaining operable LPCI pumps shall have been or shall be demonstrated to be operable within 24 hours.</td><td>Delete CTS 4.5.A.3 and add the word, “Deleted” in its place.</td></tr>
<tr><td colspan="2">Basis / Safety Assessment:

The LPCI system is required in conjunction with the core spray system to provide adequate core cooling to the reactor core in the event of a loss of coolant accident (LOCA). The LPCI subsystems are completely independent of the CS systems; however, LPCI does function with CS to prevent excessive fuel clad temperature in accordance with the requirements of 10CFR50, Appendix K.

Should one LPCI pump become inoperable, the remaining active components of the LPCI system and all active components of both core spray subsystems, and the EDGs are required to be operable by Limiting Condition for Operation 3.5.A.3. These systems provide extensive margin to that needed to ensure adequate core cooling. CTS 3.5.A.3 only permits continued reactor operation for up to 7 days with one LPCI pump inoperable, provided that the conditions of CTS 3.5.A.3 are met.

Current SR 4.5.A.1, which includes pump testing in accordance with Section XI of the ASME code, provides adequate surveillance to ensure operability of the redundant subsystem. Alternate train pump testing, which currently must occur within 24 hours of a pump being declared inoperable, does not contribute any significant assurance of system operability, and may in fact lead to unintended or premature pump failure.

This change is therefore acceptable, since adequate assurance of LPCI pump operational readiness is provided through other SRs.</td></tr>
</table>

Table 2
(continued)

Change #	Current Technical Specification	Proposed Change
5	CTS 4.5.A.4 requires that when a LPCI subsystem is made or found to be inoperable, the active components of the redundant LPCI subsystem shall have been or shall be demonstrated to be operable within 24 hours (except the recirculation system discharge valves).	Delete CTS 4.5.A.4 and add the word, "Deleted" in its place.
	Basis / Safety Assessment: The LPCI system is required in conjunction with the core spray system to provide adequate core cooling to the reactor core in the event of a loss of coolant accident (LOCA). The LPCI subsystems are completely independent of the core spray system; however, LPCI does function with CS to prevent excessive fuel clad temperature in accordance with the requirements of 10CFR50, Appendix K. Should one LPCI subsystem become inoperable, the remaining active components of the LPCI system and all active components of both core spray subsystems, and the EDGs are required to be operable by Limiting Condition for Operation 3.5.A.4. These systems provide extensive margin to that needed to ensure adequate core cooling. CTS 3.5.A.4 only permits continued reactor operation for up to 7 days with one LPCI subsystem inoperable, provided that the conditions of CTS 3.5.A.4 are met. Current SR 4.5.A.1, which includes testing in accordance with Section XI of the ASME code, provides adequate surveillance to ensure operability of the redundant subsystem. Alternate train testing, which currently must occur within 24 hours of a subsystem being declared inoperable, does not contribute any significant assurance of system operability, and may in fact lead to unintended or premature system failure. This change is therefore acceptable, since adequate assurance of LPCI system operational readiness is provided through other SRs.	

Table 2
(continued)

<table>
<tr><th>Change #</th><th>Current Technical Specification</th><th>Proposed Change</th></tr>
<tr><td rowspan="2">6</td><td>CTS 4.5.B.2 requires that when a containment cooling subsystem is made or found to be inoperable, the active components of the redundant containment cooling subsystem shall have been or shall be demonstrated to be operable within 24 hours.</td><td>Delete CTS 4.5.B.2 and add the word, “Deleted” in its place.</td></tr>
<tr><td colspan="2">Basis / Safety Assessment:

CTS 4.5.B.2 requires that the remaining Containment Cooling subsystem be demonstrated to be operable within 24 hours of determining inoperability of the redundant Containment Cooling subsystem. This change precludes performing unnecessary testing and will allow credit to be taken for normal periodic surveillance as a verification of operability and availability of the remaining Containment Cooling subsystem. Assurance that the redundant subsystem will perform its intended safety function is obtained from the results of component testing performed in accordance with the VY Inservice Testing program, as required by SR 4.6.E.2. This periodic and post-maintenance testing is adequate to ensure system operability.

This change is acceptable because existing SR 4.5.B.2 only provides marginal assurance of system operability, and may in fact have an overall cumulative negative effect on plant safety.</td></tr>
</table>

Table 2
(continued)

Change #	Current Technical Specification	Proposed Change
7	CTS 4.5.C.2 requires that when one of the residual heat removal (RHR) service water pumps is made or found to be inoperable, the operable RHR service water pumps shall have been or shall be demonstrated to be operable within 24 hours.	Delete CTS 4.5.C.2 and add the word, "Deleted" in its place.
	Basis / Safety Assessment: CTS 4.5.C.2 requires that the remaining RHR service water pumps be demonstrated to be operable within 24 hours of determining inoperability of one RHR service water pump. This change precludes performing unnecessary pump testing and will allow credit to be taken for normal periodic surveillance as a verification of operability and availability of the remaining RHR service water pumps. Assurance that the other RHR service water pumps will perform their intended safety function is obtained from the results of the testing performed in accordance with the VY IST program, as required by SR 4.5.C.1 and SR 4.6.E.2. This periodic and post-maintenance testing is adequate to ensure system operability. This change is acceptable because existing SR 4.5.C.2 only provides marginal assurance of system operability, and may in fact have an overall cumulative negative effect on plant safety.	

Table 2
(continued)

Change #	Current Technical Specification	Proposed Change
8	CTS 4.5.C.3 requires that when one RHR service water subsystem is made or found to be inoperable, the active components of the redundant RHR service water subsystem shall have been or shall be demonstrated to be operable within 24 hours.	Delete CTS 4.5.C.3 and add the word, "Deleted" in its place.
	Basis / Safety Assessment: CTS 4.5.C.3 requires that the active components of the redundant RHR service water subsystem be demonstrated to be operable within 24 hours of determining inoperability of one RHR service water subsystem. This change precludes performing unnecessary component testing and will allow credit to be taken for normal periodic surveillance as a verification of operability and availability of the remaining RHR service water subsystem. Assurance that the active components of the redundant RHR service water subsystem will perform their intended safety function is obtained from the results of component testing performed in accordance with the VY IST program, as required by SR 4.6.E.2. and SR 4.5.C.1. This periodic and post-maintenance testing is adequate to ensure system operability. This change is acceptable because existing SR 4.5.C.3 only provides marginal assurance of system operability, and may in fact have an overall cumulative negative effect on plant safety.	

Table 2
(continued)

Change #	Current Technical Specification	Proposed Change
9	CTS 4.5.D.2 requires that when the station service water system is made or found to be unable to provide adequate cooling to one of the two essential equipment cooling loops, the remaining active components of the station service water system, both essential equipment cooling loops, and the alternate cooling tower fan, shall have been or shall be demonstrated to be operable within 24 hours.	Delete CTS 4.5.D.2 and add the word, "Deleted" in its place.
	Basis / Safety Assessment: CTS 4.5.D.2 requires that the remaining active components of the station service water system, both essential equipment cooling loops, and the alternate cooling tower fan be demonstrated to be operable within 24 hours of determining that the station service water system is unable to provide adequate cooling to one of the two essential equipment loops. This change precludes performing unnecessary component testing and will allow credit to be taken for normal periodic surveillance as a verification of operability and availability of the remaining active components of the station service water system, the two essential equipment cooling loops, and the alternate cooling tower fan. Assurance that these active components will perform their intended safety function is obtained from the results of component testing performed in accordance with the VY IST program, as required by SR 4.6.E.2. and SR 4.5.D.1. This periodic and post-maintenance testing is adequate to ensure system operability. This change is acceptable because existing SR 4.5.D.2 only provides marginal assurance of system operability, and may in fact have an overall cumulative negative effect on plant safety.	

Table 2
(continued)

Change #	Current Technical Specification	Proposed Change
10	CTS 4.5.D.3 requires that when the alternate cooling tower system is made or found to be inoperable, all active components of the station service water system and both essential equipment cooling loops shall have been or shall be demonstrated to be operable within 24 hours.	Delete CTS 4.5.D.3 and add the word, "Deleted" in its place.
	Basis / Safety Assessment: CTS 4.5.D.3 requires that all active components of the station service water system and both essential equipment cooling loops be demonstrated to be operable within 24 hours of determining that the Alternate Cooling Tower System is inoperable. This change precludes performing unnecessary component testing and will allow credit to be taken for normal periodic surveillance as a verification of operability and availability of the active components of the station service water system and the two essential equipment cooling loops. Assurance that these active components will perform their intended safety function is obtained from the results of component testing performed in accordance with the VY IST program, as required by SR 4.6.E.2. and SR 4.5.D.1. This periodic and post-maintenance testing is adequate to ensure system operability. This change is acceptable because existing SR 4.5.D.2 only provides marginal assurance of system operability, and may in fact have an overall cumulative negative effect on plant safety.	

Table 2
(continued)

Change #	Current Technical Specification	Proposed Change
11	CTS 4.5.E.2 requires that when the high pressure coolant injection (HPCI) subsystem is made or found to be inoperable, the automatic depressurization system shall have been or shall be demonstrated to be operable within 24 hours.	Delete CTS 4.5.E.2 and add the word, "Deleted" in its place.
	Basis / Safety Assessment: CTS 4.5.E.2 requires that the Automatic Depressurization System (ADS) be demonstrated to be operable within 24 hours of determining that the HPCI system is inoperable. This change precludes performing unnecessary component testing and will allow credit to be taken for normal periodic surveillance as a verification of operability and availability of the ADS. Assurance that the ADS will perform its intended safety function is obtained from the results of component testing performed in accordance with the VY IST program, as required by SR 4.6.E.2. and SR 4.5.F.1. This periodic and post-maintenance testing is adequate to ensure system operability. This change is acceptable because existing SR 4.5.E.2 only provides marginal assurance of system operability, and may in fact have an overall cumulative negative effect on plant safety.	

Table 2
(continued)

<table>
<tr><th>Change #</th><th>Current Technical Specification</th><th>Proposed Change</th></tr>
<tr><td rowspan="2">12</td><td>CTS 4.5.F.2 requires that when one relief valve of the automatic pressure relief system is made or found to be inoperable, the HPCI subsystem shall have been or shall be demonstrated to be operable within 24 hours.</td><td>Delete CTS 4.5.F.2 and add the word, "Deleted" in its place.</td></tr>
<tr><td colspan="2">Basis / Safety Assessment:

CTS 4.5.F.2 requires that the HPCI System be demonstrated to be operable within 24 hours of determining that one relief valve of the automatic pressure relief subsystem is inoperable. This change precludes performing unnecessary component testing and will allow credit to be taken for normal periodic surveillance as a verification of operability and availability of the HPCI system. Assurance that HPCI will perform its intended safety function is obtained from the results of component testing performed in accordance with the VY IST program, as required by SR 4.6.E.2. and SR 4.5.E.1. This periodic and post-maintenance testing is adequate to ensure system operability.

This change is acceptable because existing SR 4.5.F.2 only provides marginal assurance of system operability, and may in fact have an overall cumulative negative effect on plant safety.</td></tr>
</table>

Table 2
(continued)

Change #	Current Technical Specification	Proposed Change
13	CTS 4.5.H.1 requires that when one of the emergency diesel generators (EDGs) is made or found to be inoperable, the remaining EDG shall have been or shall be demonstrated to be operable within 24 hours.	Delete CTS 4.5.H.1 and add the word, "Deleted" in its place.
	Basis / Safety Assessment: The statement in CTS 4.5.H.1 is unnecessary and redundant to TS 4.10.B.1. See change #15 below for a further modification to TS 4.10.B.1. Since CTS 4.5.H.1 and CTS 4.10.B.1consider the same condition, and TS 4.5.H.1 only serves to repeat required actions, the latter is superfluous and only adds unnecessary duplication within the TS. Besides, it is more appropriate to locate EDG testing requirements in TS 4.10 than TS 4.5. Since TS 4.5.H.1 adds no additional technical or administrative requirements to the TS, it may be deleted. This is considered an administrative change and is acceptable since it removes unnecessary and redundant requirements from TS.	
14	CTS 4.7.B.3.c requires that when one train of the standby gas treatment system is made or found to be inoperable, the other train shall have been or shall be demonstrated to be operable within 24 hours.	Delete CTS 4.7.B.3.c and add the word, "Deleted" in its place.
	Basis / Safety Assessment: This change precludes unnecessary testing of the standby gas treatment system and allows credit to be taken for normal periodic surveillances as a verification of operability and availability of the remaining standby gas treatment (SGT) subsystem when one is determined to be inoperable. Assurance that the redundant train of SGT will perform its intended safety function is obtained from the results of testing performed in accordance with the SR 4.7.B. This testing is comprehensive and is adequate to ensure system operability. This change is acceptable because existing SR 4.7.B.3.c only provides marginal assurance of system operability, and may in fact have an overall cumulative negative effect on plant safety.	

Table 2
(continued)

Change #	Current Technical Specification	Proposed Change
15	CTS 4.10.B.1 specifies that when one of the diesel generators is made or found to be inoperable, the requirements of Specification 4.5.H.1 shall be satisfied.	Revise TS 4.10.B.1 to require: *When one of the emergency diesel generators is made or found to be inoperable:* *a. Within 24 hours determine that the remaining diesel generator is not inoperable due to common cause failure; or* *b. The remaining diesel generator shall have been or shall be demonstrated to be operable within 24 hours.*

Table 2
(continued)

15	**Basis / Safety Assessment:** The requirement for EDG testing when one of the EDGs is made or found to be inoperable is currently contained in CTS 4.5.H.1. This requirement is being relocated to TS 4.10.B.1 and further modified as described herein. (See also change #13 above.) CTS 4.5.H.1 requires that the remaining EDG be demonstrated to be operable within 24 hours of determining that one EDG is inoperable. This modification of the requirement precludes performing unnecessary EDG testing and will allow credit to be taken for normal periodic surveillance as a verification of operability and availability of the EDGs when it can be determined within 24 hours that the cause of the failure of the first EDG does not exist for the other EDG. If this condition is met, assurance that the remaining EDG will perform its intended safety function is obtained from the results of EDG testing performed as required by SR 4.10.A. This periodic testing is adequate to ensure system operability, provided that a common cause failure does not exist. This change is acceptable because, if it is known that a common cause failure of the EDG does not exist, current SR 4.5.H.1 only provides marginal assurance of system operability, and may in fact have an overall cumulative negative effect on plant safety. Proposed TS 4.10.B.1 provides an allowance to avoid unnecessary testing of the operable EDG. If it can be determined that the cause of the inoperable EDG (e.g., removal from service to perform routine maintenance or testing) does not exist on the operable EDG, demonstration of operability of the remaining EDG does not have to be performed. If the cause of inoperability exists on the remaining EDG, it is declared inoperable upon discovery, and Limiting Condition for Operation 3.5.H.1 requires reactor shutdown within 24 hours. Once the failure is repaired, and the common cause failure no longer exists, revised TS 4.10.B.1 is satisfied. If the cause of the initial inoperable EDG cannot be confirmed not to exist on the remaining EDG, performance of new SR 4.10.B.1.b suffices to provide assurance of continued operability of that EDG. In the event the inoperable EDG is restored to operable status prior to completing either new SR 4.10.B.1.a or new SR 4.10.B.1.b, the plant corrective action program will continue to evaluate the common cause possibility. This continued evaluation, however, is no longer under the 24 hour constraint imposed while in the condition of SR 4.10.B.1. According to NRC Generic Letter 84-15, 24 hours is a reasonable time to confirm that the operable EDG is not affected by the same problem as the inoperable EDG. These clarifications are included in the revised TS Bases and are consistent with STS. This change is therefore acceptable because it provides adequate assurance of operability of the remaining EDG when one EDG is made or found to be inoperable.

Table 2
(continued)

<table>
<tr><th>Change #</th><th>Current Technical Specification</th><th>Proposed Change</th></tr>
<tr><td rowspan="2">16</td><td>CTS 4.10.B.3.b.2 requires that the remaining diesel generator shall have been or shall be demonstrated to be operable within 24 hours when either off-site power source and one diesel or associated buses are unavailable.</td><td>Revise TS 4.10.B.3.b.2 in its entirety to:

The requirements of Specification 4.10.B.1 shall be met within 24 hours.</td></tr>
<tr><td colspan="2">Basis / Safety Assessment:

CTS 4.10.B.3.b.2 requires that the remaining EDG be demonstrated to be operable within 24 hours of determining that either off-site power source and one EDG or associated buses are unavailable. This change precludes performing unnecessary EDG testing and will allow credit to be taken for normal periodic surveillance as a verification of operability and availability of the EDGs when it can be determined within 24 hours that the cause of the failure of the first EDG does not exist for the other EDG (see Change #15 above). If new TS 4.10.B.1 is met, assurance that the remaining EDG will perform its intended safety function is obtained from the results of EDG testing performed as required by SR 4.10.A. This periodic testing is adequate to ensure system operability when it can be determined that a common cause failure does not exist.

This change is acceptable because, if it is known that a common cause failure of the EDG does not exist, current SR 4.5.H.1 only provides marginal assurance of system operability, and may in fact have an overall cumulative negative effect on plant safety.

Revised TS 4.10.B.3.b.2 provides an allowance to avoid unnecessary testing of the operable EDG. If it can be determined that the cause of the inoperable EDG (e.g., removal from service to perform routine maintenance or testing) does not exist on the operable EDG, demonstration of operability of the remaining EDG does not have to be performed. If common cause of EDG inoperability is not or can not be determined not to exist within 24 hours, the remaining EDG shall be demonstrated to be operable within the same 24 hour period. If both EDGs are declared inoperable, Limiting Condition for Operation 3.5.H.1 requires reactor shutdown within 24 hours of determination.

This change is therefore acceptable because it provides adequate assurance of operability of the remaining EDG when one EDG or its associated bus is made or found to be inoperable.</td></tr>
</table>

Table 2
(continued)

Change #	Current Technical Specification	Proposed Change
17	The TS Bases provide explanation and rationale for associated TS requirements and how they are to be implemented.	Associated changes to the TS Bases are being made to conform to the changed TS and to add clarity to existing requirements.
	Basis / Safety Assessment: This proposed change revises the Bases of the VY TSs to accurately reflect the proposed TS changes and incorporate some of assessments presented herein. Other Bases changes are made for clarity purposes and to conform to the changes being made to the associated Specifications. Bases do not establish actual requirements, and as such do not change technical requirements of the TS. The Bases changes are therefore acceptable, since they administratively document the reasons and additional understanding for the associated TSs.	

Conclusion/Summary

VY concludes that this proposed change does not adversely affect plant safety and will result in a net benefit to the safe operation of the facility, and is therefore acceptable. Based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; (2) such activities will be conducted in compliance with the Commission's regulations; and (3) issuance of the requested license amendment will not be inimical to the common defense and security or to the health and safety of the public.

Attachment 2

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 248

Alternate Train Testing

Determination of No Significant Hazards Consideration

Description of Amendment Request:

The proposed change would revise TS requirements associated with demonstrating the operability of alternate trains when redundant equipment is made or found to be inoperable. These excessive testing requirements are unnecessary to maintain adequate assurance of system operability, and may result in unintended consequences, such as premature wear of components. Other surveillance requirements provide adequate assurance that redundant systems and components are operable and capable of performing their intended safety function.

Additional changes are included that are administrative in nature and do not affect technical requirements. Associated changes to the TS Bases are also being made to conform to the changed TS.

Basis for No Significant Hazards Determination:

Pursuant to 10CFR50.92, VY has reviewed the proposed change and concludes that the change does not involve a significant hazards consideration since the proposed change satisfies the criteria in 10CFR50.92(c). These criteria require that the operation of the facility in accordance with the proposed amendment will not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety. The discussion below addresses each of these criteria and demonstrates that the proposed amendment does not constitute a significant hazard.

1. Will the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

   Because changing surveillance test requirements does not change the probability of accident precursors, this proposed change does not affect the probability of an accident previously evaluated. Since other periodic and post-maintenance surveillance requirements ensure that the operability of systems and components is maintained, there is no significant increase in the consequences of accidents previously evaluated.

   Furthermore, the removal of the additional surveillance testing from the Technical Specifications would result in a decrease in the probability of equipment failure because the excessive testing causes unnecessary wear on the safety-related equipment and unnecessary challenges to safety systems. Reduced testing may also eliminate the potential for human error associated with system alignments and misdirection of attention from monitoring and directing plant operations.

   Administrative changes to the Technical Specifications do not alter any technical requirements, and as such, do not increase the probability or consequences of accidents.

   Therefore, the proposed change will not increase the probability or consequences of any accident previously evaluated.

2. Will the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

   Reduced surveillance testing does not create new or different kinds of accidents since modes of operation are unchanged and additional accident precursors are not introduced. System operability requirements and design bases remain the same, and reactor operations are unchanged. Since system and component testing only involves the assurance of operability, reduced testing does not introduce mechanisms that may contribute to the possibility of new or different kinds of accidents.

   Administrative changes to the Technical Specifications do not alter any technical requirements, and as such, do not create the possibility of new or different kinds of accidents.

   Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Will the proposed changes involve a significant reduction in a margin of safety?

   The proposed change will not decrease operability requirements, nor reduce the equipment required during various plant conditions. An acceptable level of testing exists in other Technical Specification requirements to demonstrate system and component operability. There are no changes to system or component operability requirements; therefore, systems and components will be available to provide existing margins of safety. The same systems and components with the same performance levels assumed in safety analyses will still be available to mitigate consequences of postulated accidents.

   Administrative changes to the Technical Specifications do not alter any technical requirements, and as such, have no effect on margins of safety.

   Therefore, this change does not involve a significant reduction in a margin of safety.

Conclusion

On the basis of the above, VY has determined that operation of the facility in accordance with the proposed change does not involve a significant hazards consideration as defined in 10CFR50.92(c), in that it: (1) does not involve a significant increase in the probability or consequences of an accident previously evaluated; (2) does not create the possibility of a new or different kind of accident from any accident previously evaluated; and (3) does not involve a significant reduction in a margin of safety.

Attachment 3

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 248

Alternate Train Testing

Marked-up Version of the Current Technical Specifications

3.4 LIMITING CONDITIONS FOR OPERATION

B. Operation with Inoperable Components

From and after the date that a redundant component is made or found to be inoperable, reactor operation is permissible during the succeeding seven days unless such component is sooner made operable.

C. Standby Liquid Control System Tank - Borated Solution

At all times when the Standby Liquid Control System is required to be operable, the following conditions shall be met:

1. The net volume versus concentration of the sodium pentaborate solution in the standby liquid control tank shall meet the requirements of Figure 3.4.1.

4.4 SURVEILLANCE REQUIREMENTS

5. Testing the new trigger assemblies by installing one of the assemblies in the test block and firing it using the installed circuitry. Install the unfired assemblies, taken from the same batch as the fired one, into the explosion valves.

6. Recirculating the borated solution.

1

Deleted.

B. Operation with Inoperable Components

~~When a component becomes inoperable, its redundant component shall be or shall have been demonstrated to be operable within 24 hours.~~

C. Standby Liquid Control System Tank - Borated Solution

1. The solution volume in the tank and temperature in the tank and suction piping shall be checked at least daily.

be inoperable, there is no immediate threat to shutdown capability, and reactor operation may continue while repairs are being made. Assurance that the system will perform its intended function is obtained from the results of the pump and valve testing performed in accordance with ASME Section XI requirements. ~~Whenever one redundant component is inoperable, the potential for extended operation with two subsystems inoperable is reduced by requiring that the redundant component is tested within 24 hours.~~

17

C. Standby Liquid Control System Tank - Borated Solution

The solution saturation temperature varies with the concentration of sodium pentaborate. The solution shall be kept at least 10°F above the saturation temperature to guard against boron precipitation. The 10°F margin is included in Figure 3.4.2. Temperature and liquid level alarms for the system are annunciated in the Control Room.

Once the solution has been made up, boron concentration will not vary unless more boron or water is added. Level indication and alarm indicate whether the solution volume has changed which might indicate a possible solution concentration change. Considering these factors, the test interval has been established.

Sodium pentaborate concentration is determined within 24 hours following the addition of water or boron, or if the solution temperature drops below specified limits. The 24-hour limit allows for 8 hours of mixing, subsequent testing, and notification of shift personnel.

Boron concentration, solution temperature, and volume are checked on a frequency to assure a high reliability of operation of the system should it ever be required. Isotopic tests of the sodium pentaborate are performed periodically to ensure that the proper boron-10 atom percentage is being used.

10CFR50.62(c)(4) requires a Standby Liquid Control System with a minimum flow capacity and boron content equivalent to 86 gpm of 13 weight percent natural sodium pentaborate solution in the 251-inch reactor pressure vessel reference plant. Natural sodium pentaborate solution is 19.8 atom percent boron-10. The relationship expressed in Specification 3.4.C.3 also contains the ratio M251/M to account for the difference in water volume between the reference plant and Vermont Yankee. (This ratio of masses is 628,300 lbs./401,247 lbs.)

To comply with the ATWS rule, the combination of three Standby Liquid Control System parameters must be considered: boron concentration, Standby Liquid Control System pump flow rate, and boron-10 enrichment. Fixing the pump flow rate in Specification 3.4.C.3 at the minimum flow rate of 35 gpm conservatively establishes a system parameter that can be used in satisfying the ATWS requirement, as well as the original system design basis. If the product of the expression in Specification 3.4.C.3 is equal to or greater than unity, the Standby Liquid Control System satisfies the requirements of 10CFR50.62(c)(4).

3.5 LIMITING CONDITION FOR OPERATIONS

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability:

Applies to the operational status of the Emergency Cooling Subsystems.

Objective:

To assure adequate cooling capability for heat removal in the event of a loss-of-coolant accident or isolation from the normal reactor heat sink.

Specification:

A. Core Spray and Low Pressure Coolant Injection

1. Except as specified in Specifications 3.5.A.2 through 3.5.A.4 below and 3.5.H.3 and 3.5.H.4, both Core Spray and the LPCI Subsystems shall be operable* whenever irradiated fuel is in the reactor vessel and prior to a reactor startup from the cold shutdown condition.

*Note: During Hot Shutdown, LPCI subsystems may be considered OPERABLE during alignment and operation for decay heat removal with reactor vessel pressure less than the RHR shutdown cooling permissive pressure, if capable of being manually realigned and not otherwise inoperable.

4.5 SURVEILLANCE REQUIREMENT

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability: Applies 2

~~Applied~~ to periodic testing of the emergency cooling subsystems.

Objective:

To verify the operability of the core containment cooling subsystems.

Specification:

A. Core Spray and Low Pressure Cooling Injection

Surveillance of the Core Spray and LPCI Subsystems shall be performed as follows.

1. General Testing

Item	Frequency
a. Simulated Automatic Actuation Test	Each re-fueling outage
b. Operability testing of pumps and valves shall be in accordance with Specification 4.6.E.	
c. Flow Rate Test-Core Spray pumps shall deliver at least 3000 gpm (torus to torus) against a system head of 120 psig. Each LPCI pump shall deliver 7450 ± 150 gpm (vessel to vessel).	Each re-fueling outage

3.5 LIMITING CONDITION FOR OPERATION

2. From and after the date that one of the Core Spray Subsystems is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such subsystem is sooner made operable, provided that during such seven days, all active components of the other Core Spray Subsystem, the LPCI Subsystems, and the diesel generators required for operation of such components if no external source of power were available, shall be operable.

3. From and after the date that one of the LPCI pumps is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such pump is sooner made operable, provided that during such seven days, the remaining active components of the LPCI Containment Cooling Subsystem and all active components of both Core Spray Subsystems and the diesel generators required for operation of such components if no external source of power were available, shall be operable.

4.5 SURVEILLANCE REQUIREMENT

2. ~~When one Core Spray Subsystem is made or found to be inoperable, the active components of the redundant Core Spray Subsystem shall have been or shall be demonstrated to be operable within 24 hours.~~ 3

Deleted.

3. ~~When one of the LPCI pumps is made or found to be inoperable, the remaining operable LPCI pumps shall have been or shall be demonstrated to be operable within 24 hours.~~ 4

3.5 LIMITING CONDITION FOR OPERATION

4. a. From and after the date that a LPCI Subsystem is made or found to be inoperable due to failure of the associated UPS, reactor operation is permissible only during the succeeding thirty days, for the 1989/90 operating cycle, unless it is sooner made operable, provided that during that time the associated motor control center (89A or 89B) is powered from its respective maintenance tie, all active components of the other LPCI and the Containment Cooling Subsystem, the Core Spray Subsystems, and the emergency diesel generators shall be operable, the requirements of Specification 3.10.A.4 are met, and the 4160 volt tie line to the Vernon Hydro is the operable delayed access power source.

b. From and after the date that a LPCI Subsystem is made or found to be inoperable for any reason, other than failure of the UPS during the 1989/90 operating cycle, or Specification 3.5.A.4.a is not met, reactor operation is permissible only during the succeeding seven days unless it is sooner made operable, provided

4.5 SURVEILLANCE REQUIREMENT

5

4. ~~When a LPCI Subsystem is made or found to be inoperable, the active components of the redundant LPCI Subsystem shall have been or shall be demonstrated to be operable within 24 hours (except the Recirculation System discharge valves).~~

Deleted.

3.5 LIMITING CONDITION FOR OPERATION

that during that time all active components of the other LPCI and the Containment Cooling Subsystem, the Core Spray Subsystems, and the diesel generators required for operation of such components if no external source of power were available, shall be operable.

5. All recirculation pump discharge valves and bypass valves shall be operable or closed prior to reactor startup.

6. If the requirements of Specifications 3.5.A cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

B. <u>Containment Spray Cooling Capability</u>

1. Both containment cooling spray loops are required to be operable when the reactor water temperature is greater than 212°F except that a Containment Cooling Subsystem may be inoperable for thirty days.

2. If this requirement cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

4.5 SURVEILLANCE REQUIREMENT

5. Recirculation pump discharge valves shall be tested to verify full open to full closed in $27 \leq t \leq 33$ seconds and bypass valves shall be tested for operability in accordance with Specification 4.6.E.

B. <u>Containment Spray Cooling Capability</u>

1. Surveillance of the drywell spray loops shall be performed as follows. During each five-year period, an air test shall be performed on the drywell spray headers and nozzles.

2. ~~When a Containment Cooling Subsystem is made or found to be inoperable, the active components of the redundant Containment Cooling Subsystem shall have been or shall be demonstrated to be operable within 24 hours.~~

6

Deleted.

3.5 LIMITING CONDITION FOR OPERATION

C. Residual Heat Removal (RHR) Service Water System

1. Except as specified in Specifications 3.5.C.2, and 3.5.C.3 below, both RHR Service Water Subsystem loops shall be operable whenever irradiated fuel is in the reactor vessel and prior to reactor startup from a cold condition.

2. From and after the date that one of the RHR service water pumps is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding thirty days unless such pump is sooner made operable, provided that during such thirty days all other active components of the RHR Service Water Subsystem are operable.

3. From and after the date that one RHR Service Water Subsystem is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such subsystem is sooner made operable, provided that all active components of the other RHR Service Water

4.5 SURVEILLANCE REQUIREMENT

C. Residual Heat Removal (RHR) Service Water System

Surveillance of the RHR Service Water System shall be performed as follows:

1. RHR Service Water Subsystem testing:

Operability testing of pumps and valves shall be in accordance with Specification 4.6.E.

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2. ~~When one of the RHR service water pumps is made or found to be inoperable, the operable RHR service water pumps shall have been or shall be demonstrated to be operable within 24 hours.~~

Deleted.

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3. ~~When one RHR Service Water Subsystem is made or found to be inoperable, the active components of the redundant RHR Service Water Subsystem shall have been or shall be demonstrated to be operable within 24 hours.~~

3.5 LIMITING CONDITION FOR OPERATION

Subsystem, both Core Spray Subsystems, and both diesel generators required for operation of such components if no external source of power were available, shall be operable.

4. If the requirements of Specification 3.5.C cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

D. <u>Station Service Water and Alternate Cooling Tower Systems</u>

1. Except as specified in Specifications 3.5.D.2 and 3.5.D.3, the Station Service Water System and both essential equipment cooling loops and the alternate cooling tower shall be operable whenever irradiated fuel is in the reactor vessel and reactor coolant temperature is greater than 212°F.

2. From and after the date that the Station Service Water System is made or found to be unable to provide adequate cooling to one of the two essential equipment cooling loops, reactor operation is permissible only during the succeeding 15 days unless adequate cooling capability to both essential equipment cooling loops is restored sooner, provided that during such 15 days all other active components of the remaining essential equipment cooling loop and the Station Service Water and Alternate Cooling Tower Systems are operable.

4.5 SURVEILLANCE REQUIREMENT

D. <u>Station Service Water and Alternate Cooling Tower Systems</u>

Surveillance of the Station Service Water and Alternate Cooling Tower Systems shall be performed as follows:

1. Operability testing of pumps and valves shall be in accordance with Specification 4.6.E.

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2. ~~When the Station Service Water System is made or found to be unable to provide adequate cooling to one of the two essential equipment cooling loops, the remaining active components of the Station Service Water System, both essential equipment cooling loops, and the alternate cooling tower fan, shall have been or shall be demonstrated to be operable within 24 hours.~~

Deleted.

3.5 LIMITING CONDITION FOR OPERATION

3. From and after the date that the Alternate Cooling Tower System is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days, unless the Alternate Cooling Tower System is made operable, provided that during such seven days all active components of the Station Service Water System and both essential equipment cooling loops are operable.

4. If the requirements of Specification 3.5.D cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

E. High Pressure Cooling Injection (HPCI) System

1. Except as specified in Specification 3.5.E.2, whenever irradiated fuel is in the reactor vessel and reactor steam pressure is greater than 150 psig:

   a. The HPCI System shall be operable.

   b. The condensate storage tank shall contain at least 75,000 gallons of condensate water.

4.5 SURVEILLANCE REQUIREMENT

3. ~~When the Alternate Cooling Tower System is made or found to be inoperable, all active components of the Station Service Water System and both essential equipment cooling loops shall have been or shall be demonstrated to be operable within 24 hours.~~

10

Deleted.

E. High Pressure Coolant Injection (HPCI) System

Surveillance of HPCI System shall be performed as follows:

1. Testing

   a. A simulated automatic actuation test of the HPCI System shall be performed during each refueling outage.

   b. Operability testing of the pump and valves shall be in accordance with Specification 4.6.E.

   c. Upon reactor startup, HPCI operability testing shall be performed as required by Specification 4.6.E within 24 hours after exceeding 150 psig reactor steam pressure.

3.5 LIMITING CONDITION FOR OPERATION

2. From and after the date that the HPCI Subsystem is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such subsystem is sooner made operable, provided that during such seven days all active components of the Automatic Depressurization Subsystems, the Core Spray Subsystems, the LPCI Subsystems, and the RCIC System are operable.

3. If the requirements of either Specification 3.5.E or Specification 4.5.E.1.c cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to ≤ 150 psig within 24 hours.

F. <u>Automatic Depressurization System</u>

1. Except as specified in Specification 3.5.F.2 below, the entire Automatic Depressurization Relief System shall be operable at any time the reactor steam pressure is above 150 psig and irradiated fuel is in the reactor vessel.

2. From and after the date that one of the four relief valves of the Automatic Depressurization Subsystem are made or found to be inoperable due to malfunction of the electrical portion of the valve when the

4.5 SURVEILLANCE REQUIREMENT

d. The HPCI System shall deliver at least 4250 gpm at normal reactor operating pressure when recirculating to the Condensate Storage Tank.

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2. ~~When the HPCI Subsystem is made or found to be inoperable, the Automatic Depressurization System shall have been or shall be demonstrated to be operable within 24 hours.~~

~~NOTE: Automatic Depressurization System operability shall be demonstrated by performing a functional test of the trip system logic.~~

Deleted.

F. <u>Automatic Depressurization System</u>

Surveillance of the Automatic Depressurization System shall be performed as follows:

1. Operability testing of the relief valves shall be in accordance with Specification 4.6.E.

Deleted.

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2. ~~When one relief valve of the Automatic Pressure Relief Subsystem is made or found to be inoperable, the HPCI Subsystem shall have been or shall be demonstrated to be operable within 24 hours.~~

3.5 LIMITING CONDITION FOR OPERATION

3. If the requirements of either Specification 3.5.G or Specification 4.5.G.1.c cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to ≤ 150 psig within 24 hours.

H. Minimum Core and Containment Cooling System Availability

1. During any period when one of the emergency diesel generators is inoperable, continued reactor operation is permissible only during the succeeding seven days, provided that all of the LPCI, Core Spray and Containment Cooling Subsystems connecting to the operable diesel generator shall be operable. If this requirement cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

2. Any combination of inoperable components in the Core and Containment Cooling Systems shall not defeat the capability of the remaining operable components to fulfill the core and containment cooling functions.

3. When irradiated fuel is in the reactor vessel and the reactor is in either a refueling or cold shutdown condition, all Core and Containment Cooling Subsystems may be inoperable provided no work is permitted which has the potential for draining the reactor vessel.

4.5 SURVEILLANCE REQUIREMENT

d.

2

~~c.~~ The RCIC System shall deliver at least 400 gpm at normal reactor operating pressure when recirculating to the Condensate Storage Tank.

H. Minimum Core and Containment Cooling System Availability

13

1. ~~When one of the emergency diesel generators is made or found to be inoperable, the remaining diesel generator shall have been or shall be demonstrated to be operable within 24 hours.~~

Deleted.

BASES:

3.5 CORE AND CONTAINMENT COOLANT SYSTEMS

A. Core Spray Cooling System and Low Pressure Coolant Injection System

This Specification assures that adequate standby cooling capability is available whenever irradiated fuel is in the Reactor Vessel.

Based on the loss-of-coolant analyses, the Core Spray and LPCI Systems provide sufficient cooling to the core to dissipate the energy associated with the loss-of-coolant accident and to limit the accident-caused core conditions as specified in 10CFR50, Appendix K. The analyses consider appropriate combinations of the two Core Spray Subsystems and the two LPCI Subsystems associated with various break locations and equipment availability in accordance with required single failure assumptions. (Each LPCI Subsystem consists of the LPCI pumps, the recirculation pump discharge valve, and the LPCI injection valve which combine to inject torus water into a recirculation loop.)

The LPCI System is designed to provide emergency cooling to the core by flooding in the event of a loss-of-coolant accident. This system is completely independent of the Core Spray System; however, it does function in combination with the Core Spray System to prevent excessive fuel clad temperature. The LPCI and the Core Spray Systems provide adequate cooling for break areas up to and including the double-ended recirculation line break without assistance from the high pressure emergency Core Cooling Subsystems.

Specification 3.5.A.1 is modified by a Note that allows LPCI subsystems to be considered OPERABLE during alignment and operation for decay heat removal with reactor pressure less than the RHR shutdown cooling permissive pressure, if capable of being manually realigned (remote) to the LPCI mode and not otherwise inoperable. This allows operation in the RHR shutdown cooling mode during Hot Shutdown, if necessary.

The intent of these specifications is to prevent startup from the cold condition without all associated equipment being operable. However, during operation, certain components may be out of service for the specified allowable repair times. Assurance that the systems will perform their intended function is obtained from the results of the pump and valve testing performed in accordance with ASME Section XI requirements referenced in Specification 4.6.E. ~~Whenever one redundant system is inoperable, the potential for extended operation with two subsystems inoperable is reduced by requiring that the redundant subsystem be tested within 24 hours.~~

17

B. and C. Containment Spray Cooling Capability and RHR Service Water System

The containment heat removal portion of the RHR System is provided to remove heat energy from the containment in the event of a loss-of-coolant accident. For the flow specified, the containment long-term pressure is limited to less than 5 psig and, therefore, the flow is more than ample to provide the required heat removal capability. Reference: Section 14.6.3.3.2 FSAR.

Each Containment Cooling Subsystem consists of two RHR service water pumps, 1 heat exchanger, and 2 RHR (LPCI) pumps. Either set of equipment is capable of performing the containment cooling function. In fact, an analysis in Section 14.6 of the FSAR shows that one subsystem consisting of 1 RHR service water pump, 1 heat exchanger, and 1 RHR pump has sufficient capacity to perform the cooling function. Assurance that the systems will perform their intended function is obtained from the results of the pump and valve testing performed in accordance with ASME Section XI requirements referenced in Specification 4.6.E. ~~Whenever one redundant system is inoperable, the potential for extended operation with two subsystems inoperable is reduced by requiring that the redundant subsystem be tested within 24 hours.~~

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BASES: 3.5 (Cont'd)

D. Station Service Water and Alternate Cooling Tower Systems

The Station Service Water System consists of pumps, valves and associated piping necessary to supply water to two essential equipment cooling loops and additional essential and nonessential equipment cooling loads. Each of the two Station Service Water essential equipment cooling loops includes valves, piping and associated instrumentation necessary to provide a flowpath to essential equipment. The Station Service Water essential equipment cooling loops provide redundant heat sinks to dissipate residual heat after a shutdown or accident. Each Station Service Water essential equipment cooling loop provides sufficient heat sink capacity to perform the required heat dissipation. Analyses have shown that any two service water pumps are capable of providing adequate cooling capability to the essential equipment cooling loops. To ensure this capability, four Service Water pumps and two Service Water essential equipment cooling loops must be operable. This ensures that at least two operable Service Water Pumps and one operable essential equipment cooling loop will be available in the event of the worst single active failure occurring coincident with a loss of off-site power. A Service Water pump is considered operable when it is capable of taking suction from an intake bay and transferring water to a Service Water essential equipment cooling loop at the specified pressures and flow rates. An essential equipment cooling loop is considered operable when it has a flow path capable of transferring water to the essential equipment, when required. The Alternate Cooling Tower System will provide the necessary heat sink for normal post-shutdown conditions in the event that the Station Service Water System becomes incapacitated due to a loss of the Vernon Dam with subsequent loss of the Vernon Pond, flooding of the Service Water intake structure (due to probable maximum flood in the river or an upstream dam failure) or fire in the Service Water intake structure which disables all four Service Water pumps.

If one or more Station Service Water component(s) are inoperable such that the Station Service Water System would not be capable of performing its safety function, assuming a single active failure (e.g., a pump, valve or diesel generator), then at least one essential equipment cooling loop is inoperable. If one or more component(s) are inoperable such that the Station Service Water System would not be capable of performing its safety function, even without assuming a single active failure, then both essential equipment cooling loops are inoperable.

Although the Station Service Water (SSW) System can perform its safety function with only two operable SSW pumps, the SSW System may not be capable of performing its safety function assuming one or two inoperable SSW pumps and assuming a worst case single active failure (e.g., failure of a diesel generator, SSW pump, SSW valve, etc.). Therefore, reactor operation with one or two inoperable SSW pumps is limited to 15 days provided that during this time both the normal and emergency power supplies for the remaining operable SSW pumps are also operable, in addition to ~~demonstrating~~ the operability of all remaining active components of the SSW system which perform a safety function and the alternate cooling tower fan.

If the SSW System would not be capable of performing its safety function for a reason other than one or two SSW pumps being inoperable, assuming a worst case single active failure (e.g., failure of a diesel generator,

requiring

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requiring 17

SSW pump, SSW valve, etc.), then reactor operation is limited to 15 days provided that during this time both the normal and emergency power supplies for the remaining operable equipment are also operable, in addition to ~~demonstrating~~ the operability of all remaining active components of the SSW system which perform a safety function and the alternate cooling tower fan.

If the SSW System would not be capable of performing its safety function for any reason, even without assuming a worst case single active failure, then the reactor must be placed in the cold shutdown condition within 24 hours.

E. <u>High Pressure Coolant Injection System</u>

The High Pressure Coolant Injection System (HPCIs) is provided to adequately cool the core for all pipe breaks smaller than those for which the LPCI or Core Spray Cooling Subsystems can protect the core.

The HPCIs meets this requirement without the use of outside power. For the pipe breaks for which the HPCIs is intended to function the core never uncovers and is continuously cooled; thus, no clad damage occurs and clad temperatures remain near normal throughout the transient. Reference: Subsection 6.5.2.2 of the FSAR.

F. <u>Automatic Depressurization System</u>

The Automatic Depressurization System (ADS) consists of the four safety-relief valves and serves as a backup to the High Pressure Coolant Injection System (HPCI). ADS is designed to provide depressurization of the reactor coolant system during a small break loss-of-coolant accident if HPCI fails or is unable to maintain sufficient reactor water level. Since HPCI operability is required above 150 psig, ADS operability is also required above this pressure.

ADS operation reduces the reactor pressure to within the operating pressure range of the low pressure coolant injection and core spray systems, so that these systems can provide reactor coolant inventory makeup.

G. <u>Reactor Core Isolation Cooling System</u>

The Reactor Core Isolation Cooling System (RCIC) is provided to maintain the water inventory of the reactor vessel in the event of a main steam line isolation and complete loss of outside power without the use of the emergency core cooling systems. The RCIC meets this requirement. Reference Section 14.5.4.4 FSAR. The HPCIS provides an incidental backup to the RCIC system such that in the event the RCIC should be inoperable no loss of function would occur if the HPCIS is operable.

H. <u>Minimum Core and Containment Cooling System Availability</u>

The core cooling and containment cooling subsystems provide a method of transferring the residual heat following a shutdown or accident to a heat sink. Based on analyses, this specification assures that the core and containment cooling function is maintained with any combination of allowed inoperable components.

BASES: 3.5 (Cont'd)

Operability of low pressure ECCS injection/spray subsystems is required during cold shutdown and refueling conditions to ensure adequate coolant inventory and sufficient heat removal capability for the irradiated fuel in the core in case of inadvertent draindown of the vessel. It is permissible, based upon the low heat load and other methods available to remove the residual heat, to disable all core and containment cooling systems for maintenance if the reactor is in cold shutdown or refueling and there are no operations with a potential for draining the reactor vessel (OPDRV). However, if OPDRVs are in progress with irradiated fuel in the reactor vessel, operability of low pressure ECCS injection/spray subsystems is required to ensure capability to maintain adequate reactor vessel water level in the event of an inadvertent vessel draindown. In this condition, at least 300,000 gallons of makeup water must be available to assure core flooding capability. In addition, only one diesel generator associated with one of the ECCS injection/spray subsystems is required to be operable in this condition since, upon loss of normal power supply, one ECCS subsystem is sufficient to meet this function.

The low pressure ECCS injection/spray subsystems consist of two core spray (CS) and two low pressure coolant injection (LPCI) subsystems. During cold shutdown and refueling conditions, each CS subsystem requires one motor driven pump, piping, and valves to transfer water from the suppression pool or condensate storage tank to the reactor pressure vessel (RPV). Also, during cold shutdown and refueling conditions, each LPCI subsystem requires one motor driven pump, piping, and valves to transfer water from the suppression pool to the RPV. Under these conditions, only a single LPCI pump is required per subsystem because of the larger injection capacity in relation to a CS subsystem. One LPCI subsystem may be aligned for decay heat removal and considered operable for the ECCS function, if it can be manually realigned (remote or local) to the LPCI mode and is not otherwise inoperable. Because of low pressure and low temperature conditions during cold shutdown and refueling, sufficient time will be available to manually align and initiate LPCI subsystem operation to provide core cooling prior to postulated fuel uncovery.

I. Maintenance of Filled Discharge Pipe

Full discharge lines are required when the core spray subsystems, LPCI subsystems, HPCI and RCIC are required to be operable to preclude the possibility of damage to the discharge piping due to water hammer action upon a pump start.

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During shutdown and refueling conditions, LPCI subsystems may be considered operable during RHR system alignment and operation for decay heat removal, if those subsystems are capable of being manually realigned to the LPCI mode and are not otherwise inoperable.

BASES: 4.5 (Cont'd)

The Automatic Depressurization System is tested during refueling outages to avoid an undesirable blowdown of the Reactor Coolant System.

The HPCI Automatic Actuation Test will be performed by simulation of the accident signal. The test is normally performed in conjunction with the automatic actuation of all Core Standby Cooling Systems.

G. Reactor Core Isolation Cooling System

The frequency and conditions for testing of the RCIC system are the same as for the HPCI system. Testing is conducted in accordance with Specification 4.6.E and provides assurance that the system will function as intended.

H. Minimum Core and Containment Cooling System Availability

Deleted.

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~~Assurance that the diesels will perform their intended function is obtained by the periodic surveillance test and the results obtained from the pump and valve testing performed in accordance with ASME Section XI requirements described in Specification 4.6.E. Whenever a diesel is inoperable, the potential for extended operation with two diesels inoperable is reduced by requiring that the redundant diesel be tested within 24 hours.~~

I. Maintenance of Filled Discharge Pipe

Observation of water flowing from the discharge line high point vent as ~~discussed in Section I~~ required by Specification 4.5.I assures that the Core Cooling Subsystems will not experience water hammer damage when any of the pumps are started. Core Spray Subsystems and LPCI Subsystems will also be vented through the discharge line high point vent following a return from an inoperable status to assure that the system is "solid" and ready for operation.

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3.7 LIMITING CONDITIONS FOR OPERATION

3. a. From and after the date that one train of the Standby Gas Treatment System is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such train is sooner made operable, provided that during such seven days all active components of the other standby gas treatment train shall be operable.

If this condition cannot be met during reactor operation, or the inoperable train is not restored to operable status within seven days, the actions and completion times of Specification 3.7.B.4.a shall apply.

3. b. From and after the date that one train of the Standby Gas Treatment System is made or found to be inoperable for any reason, operations requiring secondary containment are permissible during the succeeding seven days unless such train is sooner made operable, provided that during such seven days all active components, including the associated Emergency Diesel Generator of the other standby gas treatment train shall be operable.

If this condition cannot be met during a refueling or cold

4.7 SURVEILLANCE REQUIREMENTS

once per operating cycle not to exceed 18 months. If the ultrasonic test indicates the presence of a leak, the condition will be evaluated and the gasket repaired or replaced as necessary.

f. DOP and halogenated hydrocarbon test shall be performed following any design modification to the Standby Gas Treatment System housing that could have an effect on the filter efficiency.

g. An air distribution test demonstrating uniformity within ±20% across the HEPA filters and charcoal adsorbers shall be performed if the SGTS housing is modified such that air distribution could be affected.

3. a. At least once per operating cycle automatic initiation of each train of the Standby Gas Treatment System shall be demonstrated.

b. Operability testing of valves shall be in accordance with Specification 4.6.E.

c. ~~When one train of the Standby Gas Treatment System is made or found to be inoperable, the other train shall have been or shall be demonstrated to be operable within 24 hours.~~

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Deleted.

3.10 LIMITING CONDITIONS FOR OPERATION

B. Operation With Inoperable Components

Whenever the reactor is in Run Mode or Startup Mode with the reactor not in the Cold Condition, the requirements of 3.10.A shall be met except:

1. Diesel Generators

From and after the date that one of the diesel generators or its associated buses are made or found to be inoperable for any reason and the remaining diesel generator is operable, the requirements of Specification 3.5.H.1 shall be satisfied.

2. Batteries

a. From and after the date that ventilation is lost in the Battery Room portable ventilation equipment shall be provided.

b. From and after the date that one of the two 125 volt Station Battery Systems is made or found to be inoperable for any reasons, continued reactor operation is permissible only during the succeeding three days provided Specification 3.5.H is met unless such Battery System is sooner made operable.

4.10 SURVEILLANCE REQUIREMENTS

B. Operation With Inoperable Components

1. Diesel Generator

~~When one of the diesel generators is made or found to be inoperable, the requirements of Specification 4.5.H.1 shall be satisfied.~~

When one of the emergency diesel generators is made or found to be inoperable:

a. Within 24 hours determine that the remaining diesel generator is not inoperable due to common cause failure; or

b. The remaining diesel generator shall have been or shall be demonstrated to be operable within 24 hours.

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2. Batteries

Samples of the Battery Room atmosphere shall be taken daily for hydrogen concentration determination.

3.10 LIMITING CONDITIONS FOR OPERATION

b. From and after the date that either off-site power source and one diesel generator are made or found to be inoperable for any reason, continued operation is permitted for 24 hours as long as the remaining off-site power source, the remaining diesel generator, associated emergency buses and all Low Pressure Core and Containment Cooling Systems are operable.

If this requirement cannot be met, an orderly shutdown shall be initiated and the reactor shall be in cold shutdown within 24 hours.

4.10 SURVEILLANCE REQUIREMENTS

b. When either off-site power source and one diesel or associated buses are unavailable:

1. The other off-site power source and all Low Pressure Core and Containment Cooling Systems shall have been or shall be verified operable within one hour and once per eight hours thereafter.

2. ~~The remaining diesel generator shall have been or shall be demonstrated to be operable within 24 hours.~~ The requirements of Specification 4.10.B.1 shall be met within 24 hours.

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BASES: 4.10 (Cont'd)

for the associated batteries. The results of these tests will be logged and compared with the manufacturer's recommendations of acceptability.

The Service Discharge Test (4.10.A.2.c) is a test of the batteries ability to satisfy the design requirements of the associated dc system. This test will be performed using simulated or actual loads at the rates and for the durations specified in the design load profile (battery duty cycle).

Verification of operability of an off-site power source and Low Pressure Core and Containment Cooling Systems within one hour and once per eight hours thereafter as required by 4.10.B.3.b.1 may be performed as an administrative check by examining logs and other information to determine that required equipment is available and not out of service for maintenance or other reasons. It does not require performing the surveillance needed to demonstrate the operability of the equipment.

C. Logging the diesel fuel supply weekly and after each operation assures that the minimum fuel supply requirements will be maintained. During the monthly test for quality of the diesel fuel oil, a viscosity test and water and sediment test will be performed as described in ASTM D975-68. The quality of the diesel fuel oil will be acceptable if the results of the tests are within the limiting requirements for diesel fuel oils shown on Table 1 of ASTM D975-68.

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Assurance that the diesels will meet their intended function is obtained by periodic surveillance testing and the results obtained from the pump and valve testing performed in accordance with the requirements of ASME Section XI and Specification 4.6.E. Specification 4.10.B.1.a provides an allowance to avoid unnecessary testing of the operable emergency diesel generator (EDG). If it can be determined that the cause of the inoperable EDG (e.g., removal from service to perform routine maintenance or testing) does not exist on the operable EDG, demonstration of operability of the remaining EDG does not have to be performed. If the cause of inoperability exists on the remaining EDG, it is declared inoperable upon discovery, and Limiting Condition for Operation 3.5.H.1 requires reactor shutdown within 24 hours. Once the failure is repaired, and the common cause failure no longer exists, Specification 4.10.B.1.a is satisfied. If the cause of the initial inoperable EDG cannot be confirmed not to exist on the remaining EDG, performance of Surveillance Requirement (SR) 4.10.B.1.b suffices to provide assurance of continued operability of that EDG.

In the event the inoperable EDG is restored to operable status prior to completing either SR 4.10.B.1.a or SR 4.10.B.1.b, the plant corrective action program will continue to evaluate the common cause possibility. This continued evaluation, however, is no longer under the 24 hour constraint imposed while in the condition of SR 4.10.B.1 or SR 4.10.B.3.b.2.

According to NRC Generic Letter 84-15, 24 hours is a reasonable time to confirm that the operable EDG is not affected by the same problem as the inoperable EDG.

Attachment 4

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 248

Alternate Train Testing

Retyped Technical Specification Pages

Listing of Affected Technical Specifications Pages

Replace the Vermont Yankee Nuclear Power Station Technical Specifications pages listed below with the revised pages included herein. The revised pages contain vertical lines in the margin indicating the areas of change.

Remove	Insert
93	93
98	98
99	99
100	100
101	101
102	102
103	103
104	104
105	105
106	106
108	108
110	110
111	111
111a	111a
112	112
114	114
154	154
215	215
217a	217a
223	223

3.4 LIMITING CONDITIONS FOR OPERATION

B. Operation with Inoperable Components

From and after the date that a redundant component is made or found to be inoperable, reactor operation is permissible during the succeeding seven days unless such component is sooner made operable.

C. Standby Liquid Control System Tank - Borated Solution

At all times when the Standby Liquid Control System is required to be operable, the following conditions shall be met:

1. The net volume versus concentration of the sodium pentaborate solution in the standby liquid control tank shall meet the requirements of Figure 3.4.1.

4.4 SURVEILLANCE REQUIREMENTS

5. Testing the new trigger assemblies by installing one of the assemblies in the test block and firing it using the installed circuitry. Install the unfired assemblies, taken from the same batch as the fired one, into the explosion valves.

6. Recirculating the borated solution.

B. Operation with Inoperable Components

Deleted.

C. Standby Liquid Control System Tank - Borated Solution

1. The solution volume in the tank and temperature in the tank and suction piping shall be checked at least daily.

BASES: 3.4 & 4.4 (Cont'd)

be inoperable, there is no immediate threat to shutdown capability, and reactor operation may continue while repairs are being made. Assurance that the system will perform its intended function is obtained from the results of the pump and valve testing performed in accordance with ASME Section XI requirements.

C. Standby Liquid Control System Tank - Borated Solution

The solution saturation temperature varies with the concentration of sodium pentaborate. The solution shall be kept at least 10°F above the saturation temperature to guard against boron precipitation. The 10°F margin is included in Figure 3.4.2. Temperature and liquid level alarms for the system are annunciated in the Control Room.

Once the solution has been made up, boron concentration will not vary unless more boron or water is added. Level indication and alarm indicate whether the solution volume has changed which might indicate a possible solution concentration change. Considering these factors, the test interval has been established.

Sodium pentaborate concentration is determined within 24 hours following the addition of water or boron, or if the solution temperature drops below specified limits. The 24-hour limit allows for 8 hours of mixing, subsequent testing, and notification of shift personnel.

Boron concentration, solution temperature, and volume are checked on a frequency to assure a high reliability of operation of the system should it ever be required. Isotopic tests of the sodium pentaborate are performed periodically to ensure that the proper boron-10 atom percentage is being used.

10CFR50.62(c)(4) requires a Standby Liquid Control System with a minimum flow capacity and boron content equivalent to 86 gpm of 13 weight percent natural sodium pentaborate solution in the 251-inch reactor pressure vessel reference plant. Natural sodium pentaborate solution is 19.8 atom percent boron-10. The relationship expressed in Specification 3.4.C.3 also contains the ratio M251/M to account for the difference in water volume between the reference plant and Vermont Yankee. (This ratio of masses is 628,300 lbs./401,247 lbs.)

To comply with the ATWS rule, the combination of three Standby Liquid Control System parameters must be considered: boron concentration, Standby Liquid Control System pump flow rate, and boron-10 enrichment. Fixing the pump flow rate in Specification 3.4.C.3 at the minimum flow rate of 35 gpm conservatively establishes a system parameter that can be used in satisfying the ATWS requirement, as well as the original system design basis. If the product of the expression in Specification 3.4.C.3 is equal to or greater than unity, the Standby Liquid Control System satisfies the requirements of 10CFR50.62(c)(4).

3.5 LIMITING CONDITION FOR OPERATIONS

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability:

Applies to the operational status of the Emergency Cooling Subsystems.

Objective:

To assure adequate cooling capability for heat removal in the event of a loss-of-coolant accident or isolation from the normal reactor heat sink.

Specification:

A. Core Spray and Low Pressure Coolant Injection

1. Except as specified in Specifications 3.5.A.2 through 3.5.A.4 below and 3.5.H.3 and 3.5.H.4, both Core Spray and the LPCI Subsystems shall be operable* whenever irradiated fuel is in the reactor vessel and prior to a reactor startup from the cold shutdown condition.

*Note: During Hot Shutdown, LPCI subsystems may be considered OPERABLE during alignment and operation for decay heat removal with reactor vessel pressure less than the RHR shutdown cooling permissive pressure, if capable of being manually realigned and not otherwise inoperable.

4.5 SURVEILLANCE REQUIREMENT

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability:

Applies to periodic testing of the emergency cooling subsystems.

Objective:

To verify the operability of the core containment cooling subsystems.

Specification:

A. Core Spray and Low Pressure Cooling Injection

Surveillance of the Core Spray and LPCI Subsystems shall be performed as follows.

1. General Testing

Item	Frequency
a. Simulated Automatic Actuation Test	Each re-fueling outage
b. Operability testing of pumps and valves shall be in accordance with Specification 4.6.E.	
c. Flow Rate Test-Core Spray pumps shall deliver at least 3000 gpm (torus to torus) against a system head of 120 psig. Each LPCI pump shall deliver 7450 ± 150 gpm (vessel to vessel).	Each re-fueling outage

3.5 LIMITING CONDITION FOR OPERATION

2. From and after the date that one of the Core Spray Subsystems is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such subsystem is sooner made operable, provided that during such seven days, all active components of the other Core Spray Subsystem, the LPCI Subsystems, and the diesel generators required for operation of such components if no external source of power were available, shall be operable.

3. From and after the date that one of the LPCI pumps is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such pump is sooner made operable, provided that during such seven days, the remaining active components of the LPCI Containment Cooling Subsystem and all active components of both Core Spray Subsystems and the diesel generators required for operation of such components if no external source of power were available, shall be operable.

4.5 SURVEILLANCE REQUIREMENT

2. Deleted.

3. Deleted.

3.5 LIMITING CONDITION FOR OPERATION

4\. a. From and after the date that a LPCI Subsystem is made or found to be inoperable due to failure of the associated UPS, reactor operation is permissible only during the succeeding thirty days, for the 1989/90 operating cycle, unless it is sooner made operable, provided that during that time the associated motor control center (89A or 89B) is powered from its respective maintenance tie, all active components of the other LPCI and the Containment Cooling Subsystem, the Core Spray Subsystems, and the emergency diesel generators shall be operable, the requirements of Specification 3.10.A.4 are met, and the 4160 volt tie line to the Vernon Hydro is the operable delayed access power source.

b. From and after the date that a LPCI Subsystem is made or found to be inoperable for any reason, other than failure of the UPS during the 1989/90 operating cycle, or Specification 3.5.A.4.a is not met, reactor operation is permissible only during the succeeding seven days unless it is sooner made operable, provided

4.5 SURVEILLANCE REQUIREMENT

4\. Deleted.

3.5 LIMITING CONDITION FOR OPERATION

that during that time all active components of the other LPCI and the Containment Cooling Subsystem, the Core Spray Subsystems, and the diesel generators required for operation of such components if no external source of power were available, shall be operable.

5. All recirculation pump discharge valves and bypass valves shall be operable or closed prior to reactor startup.

6. If the requirements of Specifications 3.5.A cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

B. Containment Spray Cooling Capability

1. Both containment cooling spray loops are required to be operable when the reactor water temperature is greater than 212°F except that a Containment Cooling Subsystem may be inoperable for thirty days.

2. If this requirement cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

4.5 SURVEILLANCE REQUIREMENT

5. Recirculation pump discharge valves shall be tested to verify full open to full closed in $27 \leq t \leq 33$ seconds and bypass valves shall be tested for operability in accordance with Specification 4.6.E.

B. Containment Spray Cooling Capability

1. Surveillance of the drywell spray loops shall be performed as follows. During each five-year period, an air test shall be performed on the drywell spray headers and nozzles.

2. Deleted.

3.5 LIMITING CONDITION FOR OPERATION

C. Residual Heat Removal (RHR) Service Water System

1. Except as specified in Specifications 3.5.C.2, and 3.5.C.3 below, both RHR Service Water Subsystem loops shall be operable whenever irradiated fuel is in the reactor vessel and prior to reactor startup from a cold condition.

2. From and after the date that one of the RHR service water pumps is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding thirty days unless such pump is sooner made operable, provided that during such thirty days all other active components of the RHR Service Water Subsystem are operable.

3. From and after the date that one RHR Service Water Subsystem is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such subsystem is sooner made operable, provided that all active components of the other RHR Service Water Subsystem, both Core Spray Subsystems, and both diesel generators required for operation of such components if no external source of power were available, shall be operable.

4.5 SURVEILLANCE REQUIREMENT

C. Residual Heat Removal (RHR) Service Water System

Surveillance of the RHR Service Water System shall be performed as follows:

1. RHR Service Water Subsystem testing:

   Operability testing of pumps and valves shall be in accordance with Specification 4.6.E.

2. Deleted.

3. Deleted.

3.5 LIMITING CONDITION FOR OPERATION

4. If the requirements of Specification 3.5.C cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

D. Station Service Water and Alternate Cooling Tower Systems

1. Except as specified in Specifications 3.5.D.2 and 3.5.D.3, the Station Service Water System and both essential equipment cooling loops and the alternate cooling tower shall be operable whenever irradiated fuel is in the reactor vessel and reactor coolant temperature is greater than 212°F.

2. From and after the date that the Station Service Water System is made or found to be unable to provide adequate cooling to one of the two essential equipment cooling loops, reactor operation is permissible only during the succeeding 15 days unless adequate cooling capability to both essential equipment cooling loops is restored sooner, provided that during such 15 days all other active components of the remaining essential equipment cooling loop and the Station Service Water and Alternate Cooling Tower Systems are operable.

4.5 SURVEILLANCE REQUIREMENT

D. Station Service Water and Alternate Cooling Tower Systems

Surveillance of the Station Service Water and Alternate Cooling Tower Systems shall be performed as follows:

1. Operability testing of pumps and valves shall be in accordance with Specification 4.6.E.

2. Deleted.

3.5 LIMITING CONDITION FOR OPERATION

3. From and after the date that the Alternate Cooling Tower System is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days, unless the Alternate Cooling Tower System is made operable, provided that during such seven days all active components of the Station Service Water System and both essential equipment cooling loops are operable.

4. If the requirements of Specification 3.5.D cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

E. High Pressure Cooling Injection (HPCI) System

1. Except as specified in Specification 3.5.E.2, whenever irradiated fuel is in the reactor vessel and reactor steam pressure is greater than 150 psig:

   a. The HPCI System shall be operable.

   b. The condensate storage tank shall contain at least 75,000 gallons of condensate water.

4.5 SURVEILLANCE REQUIREMENT

3. Deleted.

E. High Pressure Coolant Injection (HPCI) System

Surveillance of HPCI System shall be performed as follows:

1. Testing

   a. A simulated automatic actuation test of the HPCI System shall be performed during each refueling outage.

   b. Operability testing of the pump and valves shall be in accordance with Specification 4.6.E.

   c. Upon reactor startup, HPCI operability testing shall be performed as required by Specification 4.6.E within 24 hours after exceeding 150 psig reactor steam pressure.

3.5 LIMITING CONDITION FOR OPERATION

2. From and after the date that the HPCI Subsystem is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such subsystem is sooner made operable, provided that during such seven days all active components of the Automatic Depressurization Subsystems, the Core Spray Subsystems, the LPCI Subsystems, and the RCIC System are operable.

3. If the requirements of either Specification 3.5.E or Specification 4.5.E.1.c cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to ≤ 150 psig within 24 hours.

F. <u>Automatic Depressurization System</u>

1. Except as specified in Specification 3.5.F.2 below, the entire Automatic Depressurization Relief System shall be operable at any time the reactor steam pressure is above 150 psig and irradiated fuel is in the reactor vessel.

2. From and after the date that one of the four relief valves of the Automatic Depressurization Subsystem are made or found to be inoperable due to malfunction of the electrical portion of the valve when the

4.5 SURVEILLANCE REQUIREMENT

d. The HPCI System shall deliver at least 4250 gpm at normal reactor operating pressure when recirculating to the Condensate Storage Tank.

2. Deleted.

F. <u>Automatic Depressurization System</u>

Surveillance of the Automatic Depressurization System shall be performed as follows:

1. Operability testing of the relief valves shall be in accordance with Specification 4.6.E.

2. Deleted.

3.5 LIMITING CONDITION FOR OPERATION

3. If the requirements of either Specification 3.5.G or Specification 4.5.G.1.c cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to ≤ 150 psig within 24 hours.

H. Minimum Core and Containment Cooling System Availability

1. During any period when one of the emergency diesel generators is inoperable, continued reactor operation is permissible only during the succeeding seven days, provided that all of the LPCI, Core Spray and Containment Cooling Subsystems connecting to the operable diesel generator shall be operable. If this requirement cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

2. Any combination of inoperable components in the Core and Containment Cooling Systems shall not defeat the capability of the remaining operable components to fulfill the core and containment cooling functions.

3. When irradiated fuel is in the reactor vessel and the reactor is in either a refueling or cold shutdown condition, all Core and Containment Cooling Subsystems may be inoperable provided no work is permitted which has the potential for draining the reactor vessel.

4.5 SURVEILLANCE REQUIREMENT

d. The RCIC System shall deliver at least 400 gpm at normal reactor operating pressure when recirculating to the Condensate Storage Tank.

H. Minimum Core and Containment Cooling System Availability

1. Deleted.

BASES:

3.5 CORE AND CONTAINMENT COOLANT SYSTEMS

A. Core Spray Cooling System and Low Pressure Coolant Injection System

This Specification assures that adequate standby cooling capability is available whenever irradiated fuel is in the Reactor Vessel.

Based on the loss-of-coolant analyses, the Core Spray and LPCI Systems provide sufficient cooling to the core to dissipate the energy associated with the loss-of-coolant accident and to limit the accident-caused core conditions as specified in 10CFR50, Appendix K. The analyses consider appropriate combinations of the two Core Spray Subsystems and the two LPCI Subsystems associated with various break locations and equipment availability in accordance with required single failure assumptions. (Each LPCI Subsystem consists of the LPCI pumps, the recirculation pump discharge valve, and the LPCI injection valve which combine to inject torus water into a recirculation loop.)

The LPCI System is designed to provide emergency cooling to the core by flooding in the event of a loss-of-coolant accident. This system is completely independent of the Core Spray System; however, it does function in combination with the Core Spray System to prevent excessive fuel clad temperature. The LPCI and the Core Spray Systems provide adequate cooling for break areas up to and including the double-ended recirculation line break without assistance from the high pressure emergency Core Cooling Subsystems.

Specification 3.5.A.1 is modified by a Note that allows LPCI subsystems to be considered OPERABLE during alignment and operation for decay heat removal with reactor pressure less than the RHR shutdown cooling permissive pressure, if capable of being manually realigned (remote) to the LPCI mode and not otherwise inoperable. This allows operation in the RHR shutdown cooling mode during Hot Shutdown, if necessary.

The intent of these specifications is to prevent startup from the cold condition without all associated equipment being operable. However, during operation, certain components may be out of service for the specified allowable repair times. Assurance that the systems will perform their intended function is obtained from the results of the pump and valve testing performed in accordance with ASME Section XI requirements referenced in Specification 4.6.E.

B. and C. Containment Spray Cooling Capability and RHR Service Water System

The containment heat removal portion of the RHR System is provided to remove heat energy from the containment in the event of a loss-of-coolant accident. For the flow specified, the containment long-term pressure is limited to less than 5 psig and, therefore, the flow is more than ample to provide the required heat removal capability. Reference: Section 14.6.3.3.2 FSAR.

Each Containment Cooling Subsystem consists of two RHR service water pumps, 1 heat exchanger, and 2 RHR (LPCI) pumps. Either set of equipment is capable of performing the containment cooling function. In fact, an analysis in Section 14.6 of the FSAR shows that one subsystem consisting of 1 RHR service water pump, 1 heat exchanger, and 1 RHR pump has sufficient capacity to perform the cooling function. Assurance that the systems will perform their intended function is obtained from the results of the pump and valve testing performed in accordance with ASME Section XI requirements referenced in Specification 4.6.E.

BASES: 3.5 (Cont'd)

D. Station Service Water and Alternate Cooling Tower Systems

The Station Service Water System consists of pumps, valves and associated piping necessary to supply water to two essential equipment cooling loops and additional essential and nonessential equipment cooling loads. Each of the two Station Service Water essential equipment cooling loops includes valves, piping and associated instrumentation necessary to provide a flowpath to essential equipment. The Station Service Water essential equipment cooling loops provide redundant heat sinks to dissipate residual heat after a shutdown or accident. Each Station Service Water essential equipment cooling loop provides sufficient heat sink capacity to perform the required heat dissipation. Analyses have shown that any two service water pumps are capable of providing adequate cooling capability to the essential equipment cooling loops. To ensure this capability, four Service Water pumps and two Service Water essential equipment cooling loops must be operable. This ensures that at least two operable Service Water Pumps and one operable essential equipment cooling loop will be available in the event of the worst single active failure occurring coincident with a loss of off-site power. A Service Water pump is considered operable when it is capable of taking suction from an intake bay and transferring water to a Service Water essential equipment cooling loop at the specified pressures and flow rates. An essential equipment cooling loop is considered operable when it has a flow path capable of transferring water to the essential equipment, when required. The Alternate Cooling Tower System will provide the necessary heat sink for normal post-shutdown conditions in the event that the Station Service Water System becomes incapacitated due to a loss of the Vernon Dam with subsequent loss of the Vernon Pond, flooding of the Service Water intake structure (due to probable maximum flood in the river or an upstream dam failure) or fire in the Service Water intake structure which disables all four Service Water pumps.

If one or more Station Service Water component(s) are inoperable such that the Station Service Water System would not be capable of performing its safety function, assuming a single active failure (e.g., a pump, valve or diesel generator), then at least one essential equipment cooling loop is inoperable. If one or more component(s) are inoperable such that the Station Service Water System would not be capable of performing its safety function, even without assuming a single active failure, then both essential equipment cooling loops are inoperable.

Although the Station Service Water (SSW) System can perform its safety function with only two operable SSW pumps, the SSW System may not be capable of performing its safety function assuming one or two inoperable SSW pumps and assuming a worst case single active failure (e.g., failure of a diesel generator, SSW pump, SSW valve, etc.). Therefore, reactor operation with one or two inoperable SSW pumps is limited to 15 days provided that during this time both the normal and emergency power supplies for the remaining operable SSW pumps are also operable, in addition to requiring the operability of all remaining active components of the SSW system which perform a safety function and the alternate cooling tower fan.

If the SSW System would not be capable of performing its safety function for a reason other than one or two SSW pumps being inoperable, assuming a worst case single active failure (e.g., failure of a diesel generator,

BASES: 3.5 (Cont'd)

SSW pump, SSW valve, etc.), then reactor operation is limited to 15 days provided that during this time both the normal and emergency power supplies for the remaining operable equipment are also operable, in addition to requiring the operability of all remaining active components of the SSW system which perform a safety function and the alternate cooling tower fan.

If the SSW System would not be capable of performing its safety function for any reason, even without assuming a worst case single active failure, then the reactor must be placed in the cold shutdown condition within 24 hours.

E. High Pressure Coolant Injection System

The High Pressure Coolant Injection System (HPCIs) is provided to adequately cool the core for all pipe breaks smaller than those for which the LPCI or Core Spray Cooling Subsystems can protect the core.

The HPCIs meets this requirement without the use of outside power. For the pipe breaks for which the HPCIs is intended to function the core never uncovers and is continuously cooled; thus, no clad damage occurs and clad temperatures remain near normal throughout the transient. Reference: Subsection 6.5.2.2 of the FSAR.

F. Automatic Depressurization System

The Automatic Depressurization System (ADS) consists of the four safety-relief valves and serves as a backup to the High Pressure Coolant Injection System (HPCI). ADS is designed to provide depressurization of the reactor coolant system during a small break loss-of-coolant accident if HPCI fails or is unable to maintain sufficient reactor water level. Since HPCI operability is required above 150 psig, ADS operability is also required above this pressure.

ADS operation reduces the reactor pressure to within the operating pressure range of the low pressure coolant injection and core spray systems, so that these systems can provide reactor coolant inventory makeup.

G. Reactor Core Isolation Cooling System

The Reactor Core Isolation Cooling System (RCIC) is provided to maintain the water inventory of the reactor vessel in the event of a main steam line isolation and complete loss of outside power without the use of the emergency core cooling systems. The RCIC meets this requirement. Reference Section 14.5.4.4 FSAR. The HPCIS provides an incidental backup to the RCIC system such that in the event the RCIC should be inoperable no loss of function would occur if the HPCIS is operable.

H. Minimum Core and Containment Cooling System Availability

The core cooling and containment cooling subsystems provide a method of transferring the residual heat following a shutdown or accident to a heat sink. Based on analyses, this specification assures that the core and containment cooling function is maintained with any combination of allowed inoperable components.

BASES: 3.5 (Cont'd)

Operability of low pressure ECCS injection/spray subsystems is required during cold shutdown and refueling conditions to ensure adequate coolant inventory and sufficient heat removal capability for the irradiated fuel in the core in case of inadvertent draindown of the vessel. It is permissible, based upon the low heat load and other methods available to remove the residual heat, to disable all core and containment cooling systems for maintenance if the reactor is in cold shutdown or refueling and there are no operations with a potential for draining the reactor vessel (OPDRV). However, if OPDRVs are in progress with irradiated fuel in the reactor vessel, operability of low pressure ECCS injection/spray subsystems is required to ensure capability to maintain adequate reactor vessel water level in the event of an inadvertent vessel draindown. In this condition, at least 300,000 gallons of makeup water must be available to assure core flooding capability. In addition, only one diesel generator associated with one of the ECCS injection/spray subsystems is required to be operable in this condition since, upon loss of normal power supply, one ECCS subsystem is sufficient to meet this function.

The low pressure ECCS injection/spray subsystems consist of two core spray (CS) and two low pressure coolant injection (LPCI) subsystems. During cold shutdown and refueling conditions, each CS subsystem requires one motor driven pump, piping, and valves to transfer water from the suppression pool or condensate storage tank to the reactor pressure vessel (RPV). Also, during cold shutdown and refueling conditions, each LPCI subsystem requires one motor driven pump, piping, and valves to transfer water from the suppression pool to the RPV. Under these conditions, only a single LPCI pump is required per subsystem because of the larger injection capacity in relation to a CS subsystem. During shutdown and refueling conditions, LPCI subsystems may be considered operable during RHR system alignment and operation for decay heat removal, if those subsystems are capable of being manually realigned to the LPCI mode and are not otherwise inoperable. Because of low pressure and low temperature conditions during cold shutdown and refueling, sufficient time will be available to manually align and initiate LPCI subsystem operation to provide core cooling prior to postulated fuel uncovery.

I. Maintenance of Filled Discharge Pipe

Full discharge lines are required when the core spray subsystems, LPCI subsystems, HPCI and RCIC are required to be operable to preclude the possibility of damage to the discharge piping due to water hammer action upon a pump start.

BASES: 4.5 (Cont'd)

The Automatic Depressurization System is tested during refueling outages to avoid an undesirable blowdown of the Reactor Coolant System.

The HPCI Automatic Actuation Test will be performed by simulation of the accident signal. The test is normally performed in conjunction with the automatic actuation of all Core Standby Cooling Systems.

G. Reactor Core Isolation Cooling System

The frequency and conditions for testing of the RCIC system are the same as for the HPCI system. Testing is conducted in accordance with Specification 4.6.E and provides assurance that the system will function as intended.

H. Minimum Core and Containment Cooling System Availability

Deleted.

I. Maintenance of Filled Discharge Pipe

Observation of water flowing from the discharge line high point vent as required by Specification 4.5.I assures that the Core Cooling Subsystems will not experience water hammer damage when any of the pumps are started. Core Spray Subsystems and LPCI Subsystems will also be vented through the discharge line high point vent following a return from an inoperable status to assure that the system is "solid" and ready for operation.

3.7 LIMITING CONDITIONS FOR OPERATION

3. a. From and after the date that one train of the Standby Gas Treatment System is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such train is sooner made operable, provided that during such seven days all active components of the other standby gas treatment train shall be operable.

   If this condition cannot be met during reactor operation, or the inoperable train is not restored to operable status within seven days, the actions and completion times of Specification 3.7.B.4.a shall apply.

3. b. From and after the date that one train of the Standby Gas Treatment System is made or found to be inoperable for any reason, operations requiring secondary containment are permissible during the succeeding seven days unless such train is sooner made operable, provided that during such seven days all active components, including the associated Emergency Diesel Generator of the other standby gas treatment train shall be operable.

   If this condition cannot be met during a refueling or cold

4.7 SURVEILLANCE REQUIREMENTS

once per operating cycle not to exceed 18 months. If the ultrasonic test indicates the presence of a leak, the condition will be evaluated and the gasket repaired or replaced as necessary.

f. DOP and halogenated hydrocarbon test shall be performed following any design modification to the Standby Gas Treatment System housing that could have an effect on the filter efficiency.

g. An air distribution test demonstrating uniformity within ±20% across the HEPA filters and charcoal adsorbers shall be performed if the SGTS housing is modified such that air distribution could be affected.

3. a. At least once per operating cycle automatic initiation of each train of the Standby Gas Treatment System shall be demonstrated.

   b. Operability testing of valves shall be in accordance with Specification 4.6.E.

   c. Deleted.

3.10 LIMITING CONDITIONS FOR OPERATION

B. Operation With Inoperable Components

Whenever the reactor is in Run Mode or Startup Mode with the reactor not in the Cold Condition, the requirements of 3.10.A shall be met except:

1. Diesel Generators

From and after the date that one of the diesel generators or its associated buses are made or found to be inoperable for any reason and the remaining diesel generator is operable, the requirements of Specification 3.5.H.1 shall be satisfied.

2. Batteries

a. From and after the date that ventilation is lost in the Battery Room portable ventilation equipment shall be provided.

b. From and after the date that one of the two 125 volt Station Battery Systems is made or found to be inoperable for any reasons, continued reactor operation is permissible only during the succeeding three days provided Specification 3.5.H is met unless such Battery System is sooner made operable.

4.10 SURVEILLANCE REQUIREMENTS

B. Operation With Inoperable Components

1. Diesel Generator

When one of the emergency diesel generators is made or found to be inoperable:

a. Within 24 hours determine that the remaining diesel generator is not inoperable due to common cause failure; or

b. The remaining diesel generator shall have been or shall be demonstrated to be operable within 24 hours.

2. Batteries

Samples of the Battery Room atmosphere shall be taken daily for hydrogen concentration determination.

3.10 LIMITING CONDITIONS FOR OPERATION

b. From and after the date that either off-site power source and one diesel generator are made or found to be inoperable for any reason, continued operation is permitted for 24 hours as long as the remaining off-site power source, the remaining diesel generator, associated emergency buses and all Low Pressure Core and Containment Cooling Systems are operable.

If this requirement cannot be met, an orderly shutdown shall be initiated and the reactor shall be in cold shutdown within 24 hours.

4.10 SURVEILLANCE REQUIREMENTS

b. When either off-site power source and one diesel or associated buses are unavailable:

1. The other off-site power source and all Low Pressure Core and Containment Cooling Systems shall have been or shall be verified operable within one hour and once per eight hours thereafter.
2. The requirements of Specification 4.10.B.1 shall be met within 24 hours.

BASES: 4.10 (Cont'd)

for the associated batteries. The results of these tests will be logged and compared with the manufacturer's recommendations of acceptability.

The Service Discharge Test (4.10.A.2.c) is a test of the batteries ability to satisfy the design requirements of the associated dc system. This test will be performed using simulated or actual loads at the rates and for the durations specified in the design load profile (battery duty cycle).

Assurance that the diesels will meet their intended function is obtained by periodic surveillance testing and the results obtained from the pump and valve testing performed in accordance with the requirements of ASME Section XI and Specification 4.6.E. Specification 4.10.B.1.a provides an allowance to avoid unnecessary testing of the operable emergency diesel generator (EDG). If it can be determined that the cause of the inoperable EDG (e.g., removal from service to perform routine maintenance or testing) does not exist on the operable EDG, demonstration of operability of the remaining EDG does not have to be performed. If the cause of inoperability exists on the remaining EDG, it is declared inoperable upon discovery, and Limiting Condition for Operation 3.5.H.1 requires reactor shutdown within 24 hours. Once the failure is repaired, and the common cause failure no longer exists, Specification 4.10.B.1.a is satisfied. If the cause of the initial inoperable EDG cannot be confirmed not to exist on the remaining EDG, performance of Surveillance Requirement (SR) 4.10.B.1.b suffices to provide assurance of continued operability of that EDG.

In the event the inoperable EDG is restored to operable status prior to completing either SR 4.10.B.1.a or SR 4.10.B.1.b, the plant corrective action program will continue to evaluate the common cause possibility. This continued evaluation, however, is no longer under the 24 hour constraint imposed while in the condition of SR 4.10.B.1 or SR 4.10.B.3.b.2.

According to NRC Generic Letter 84-15, 24 hours is a reasonable time to confirm that the operable EDG is not affected by the same problem as the inoperable EDG.

Verification of operability of an off-site power source and Low Pressure Core and Containment Cooling Systems within one hour and once per eight hours thereafter as required by 4.10.B.3.b.1 may be performed as an administrative check by examining logs and other information to determine that required equipment is available and not out of service for maintenance or other reasons. It does not require performing the surveillance needed to demonstrate the operability of the equipment.

C. Logging the diesel fuel supply weekly and after each operation assures that the minimum fuel supply requirements will be maintained. During the monthly test for quality of the diesel fuel oil, a viscosity test and water and sediment test will be performed as described in ASTM D975-68. The quality of the diesel fuel oil will be acceptable if the results of the tests are within the limiting requirements for diesel fuel oils shown on Table 1 of ASTM D975-68.